Docket Nos. 50-259
50-260
and 50-296

NOV 18 1978

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Tennessee Valley Authority
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Gentlemen:

The Commission has issued the enclosed Amendments Nos. 45, 41, and 18 to Facility Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. These amendments consist of changes to the Technical Specifications in response to your request of August 3, 1978 (TVA BFNP TS 113) as supplemented by letter dated October 20, 1978.

Amendment No. 18 changes the Technical Specifications to incorporate the limiting conditions for operation associated with the initial 2000 megawatt days per tonne of fuel exposure during the second fuel cycle for Unit No. 3. As agreed with your staff, TVA will submit a reanalysis of transients for the end of cycle 2 to evaluate operation of Unit No. 3 beyond 2000 MWD/t fuel exposure. These amendments also change the Technical Specifications to incorporate minor changes in the arrangements for leak testing certain primary containment isolation and check valves.

Copies of the Safety Evaluation and Notice of Issuance are also enclosed.

Sincerely,

Original Signed by
T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Amendment No. 45 to DPR-33
2. Amendment No. 41 to DPR-52
3. Amendment No. 18 to DPR-68
4. Safety Evaluation
5. Notice

See Rpts w/ltr.

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cc w/enclosures:
See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

NOVEMBER 1 8 1978

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TA Ippolito

Thomas A. Ippolito, Chief
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7812050016 1. Amendment No. 45 to DPR-33
" 2. Amendment No. 41 to DPR-52
" 3. Amendment No. 18 to DPR-68
} SEE RPTS
7812050019 4. Safety Evaluation
7812050020 5. Notice
} W/THIS LTR.

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7812050012

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 45 TO FACILITY OPERATING LICENSE NO. DPR-33

AMENDMENT NO. 41 TO FACILITY OPERATING LICENSE NO. DPR-52

AMENDMENT NO. 18 TO FACILITY OPERATING LICENSE NO. DPR-68

TENNESSEE VALLEY AUTHORITY

BROWNS FERRY NUCLEAR PLANT, UNITS NOS. 1, 2 AND 3

DOCKET NOS. 50-259, 50-260 AND 50-296

1.0 Introduction

By letter dated August 3, 1978, and supplemented by letter dated October 20, 1978, the Tennessee Valley Authority (the licensee or TVA) requested changes to the Technical Specifications (Appendix A) appended to Facility Operating Licenses Nos. DPR-33, DPR-52 and DPR-68 for the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3. The proposed amendments and revised Technical Specifications would (1) incorporate the limiting conditions for operation associated with cycle 2 operation of Unit No. 3, and (2) incorporate minor changes to the leak rate testing valve lineups to reflect the current test program being conducted in accordance with the requirements of 10 CFR 50 Appendix J.

2.0 Discussion

Browns Ferry Unit No. 3 (BF-3) shutdown on September 8, 1978 for the first refueling of this unit. During the outage, 208 of the 764 fuel assemblies were replaced. Unit No. 3 was initially fueled with 8x8 fuel assemblies manufactured by the General Electric Company (GE).

In support of the reload application, the licensee has provided the GE BWR Reload 1 licensing submittal for BF-3 (Reference 1), proposed Technical Specification changes (Reference 2), information on the BF-3 Loss of Coolant Accident (LOCA) analysis (Reference 3), and responses to NRC requests for additional information (Reference 4).

This reload involves loading of GE 8x8 fuel and GE8x8 retrofit (8x8R) fuel. The description of the nuclear and mechanical design of the 8x8 and 8x8R fuel is contained in GE's licensing topical report for BWR reloads (Reference 5). Reference 5 also contains a complete set of references to topical reports which describe GE's analytical methods for nuclear, thermal-hydraulic, transient and accident calculations, and information regarding the applicability of these methods to cores containing a mixture of 8x8 and 8x8R fuel.

Values for plant-specific data such as steady state operating pressure, core flow, safety and safety/relief valve setpoints, rated thermal power, rated steam flow, and other design parameters are provided in Reference 5. Additional plant and cycle dependent information is provided in the reload application (Reference 1), which closely follows the outline of Appendix A of Reference 5.

Reference 6 describes the staff's review, approval, and conditions of approval for the plant-specific data addressed in Reference 5. The above-mentioned plant-specific data have been used in the transient and accident analysis provided with the reload application.

Our safety evaluation (Reference 6) of the GE generic reload licensing topical report has also concluded that the nuclear and mechanical design of the 8x8R fuel, and GE's analytical methods for nuclear and thermal-hydraulic calculations as applied to mixed cores containing 8x8 and 8x8R fuel, are acceptable. Approval of the application of the analytical methods did not include plants incorporating a prompt recirculation pump trip (RPT).

Because of our review of a large number of generic considerations related to use of 8x8R fuel in mixed loadings, and on the basis of the evaluations which have been presented in Reference 6, only a limited number of additional areas of review have been included in this safety evaluation report. For evaluations of areas not specifically addressed in this safety evaluation report, the reader is referred to Reference 6.

3.0 Evaluation

3.1 Nuclear Characteristics

For Cycle 2 operation of BF-3, 208 fresh 8x8R fuel bundles of type 8DRB265 will be loaded into the core.(1) The remainder of the 764 fuel bundles in the core will be 8x8 fuel bundles of type 8D219 exposed during the previous cycle.

The fresh fuel will be loaded and the previously peripheral fuel will be shuffled inward to constitute an octant-symmetric core pattern, which is acceptable.

Based on the data provided in Sections 4 and 5 of Reference 1, both the control rod system and the standby liquid control system will have acceptable shutdown capability during Cycle 2.

3.2 Thermal-Hydraulics

3.2.1 Fuel Cladding Integrity Safety Limit

As stated in Reference 5, the minimum critical power ratio (MCPR) which may be allowed to result from core-wide or localized transients (or from undetected fuel loading errors) is 1.07. This limit has been imposed to assure that during transients 99.9% of the fuel rods will avoid transition boiling, and that transition boiling will not occur during steady state operation as a result of the worst possible fuel loading error.

The safety limit MCPR for BF-3 is being raised from 1.06 to 1.07 because the distribution of fuel rod power within the 8x8R fuel bundles is different from that of the 8x8 fuel. The reason for the difference is the presence of two rather than one water rods in 8x8R fuel. The issue has been addressed in Reference 6 and the 1.07 limit has been found acceptable for BWRs with uncertainties in flux monitoring and operational parameters no greater than those listed in Table 5-1 of Reference 5, for which the CPR distribution is within the bounds of Figures 5.2 and 5.2a of Reference 5. It has been shown in Section 5 of Reference 5 that these conditions are met for BF-3.

3.2.2 Operating Limit MCPR

Various transients or perturbations to the CPR distribution could reduce the CPR below the intended operating limit MCPR during Cycle 2 operation. The most limiting of these operational transients and the fuel loading error have been analyzed by the licensee to determine which event could potentially induce the largest reduction in the initial power ratio (ΔCPR).

The transients evaluated were the limiting pressure and power increase transient (either turbine trip or load rejection without bypass, depending on which values have the faster closure time), the limiting coolant temperature decrease transient (loss of a feedwater heater), the feedwater controller failure transient, and the control rod withdrawal error transient. Initial conditions and transient input parameters as specified in Sections 6 and 7 of Reference 1 were assumed.

The calculated systems responses and ΔCPRs for the above listed operational transients and conditions have been analyzed by the licensee. Results were as follows:

	Δ CPR 8x8	Δ CPR 8x8R
Limiting Pressure and Power Increase Transient	.14	.14
Limiting Coolant Temperature Decrease Transient	.13	.13
Feedwater Controller Failure Transient	.09	.09
Rod Withdrawal Error	.17	.14
Fuel Loading Error, Rotated Bundle*	≤.10	.10

*The misloaded bundle error is considered separately in Section 2.3.3

The above analyses include the effect of a recirculation pump trip (RPT) on turbine stop valve closure or throttle valve fast closure. This RPT feature inserts negative reactivity into the reactor due to the rapid flow decrease and resultant increased voiding. Thus, the RPT helps shut down the reactor, effectively increasing the speed of turbine-initiated scrams.

The transient analyses described above were performed with the REDY code (Reference 7). A new improved code, ODYN, has been developed by GE. The ODYN code, which uses a more physically correct model of the plant, generally predicts smaller ΔCPRs than the REDY code when the transient under study is fairly severe. However, as transient severity is lessened, ODYN predicts a greater ΔCPR than REDY (Reference 8, p. 1). Both codes are run with conservative input values, but ODYN should be a better predictor of plant behavior once these input values are specified.

GE has stated (Reference 8) that REDY can still be used because the limiting transient has a ΔCPR sufficiently large to be above the region where REDY is non-conservative with respect to ODYN. We have proceeded on this basis in approving reloads thus far.

The addition of the RPT feature to BF-3 has significantly reduced the ΔCPR associated with the limiting pressure and power increase transient. (TVA has provided no data, but we estimate a reduction in ΔCPR by roughly a factor of two based upon p. 12 of Reference 8.) This improvement has brought the BF-3 Cycle 2 transient analysis into the region where GE's assertion (Reference 8) is no longer valid. Thus, the degree of conservatism of the BF-3 Cycle 2 transient analysis must be re-evaluated.

Approximately six to eight weeks are required to reanalyze the operational transients for cycle 2 operation of Unit 3 with the ODYN code at a cost of $85,000 to $120,000. NRC has not as yet approved the ODYN code. However, the staff had requested that TVA supply an ODYN licensing basis renalaysis of the transients to compare these results with those obtained by the accepted REDY code. Initially (reference 4), TVA's position was that this renalaysis was unwarranted until such time as the ODYN code was approved by NRC.

The limited data available to the staff indicates that calculations which include axial effects and detailed steam line modeling are likely to predict more severe results than those obtained by the point kinetics REDY calculations. This possible lack of conservatism in the REDY calculations is of concern only for the end of the fuel cycle (EOC). It is known that transient severity is greatest at end-of-cycle, generally increasing by 0.06 or more in a ΔCPR during the last 2000 megawatt days per tonne (MWD/t) of fuel exposure in the cycle (section 5.2.2.5, reference 5). The transients for the Unit 3 cycle 2 reload were calculated for the EOC conditions, which are the most severe conditions. Thus, there is considerable extra conservatism in the calculated operating limit minimum critical power ratio (OLMCPR) at the beginning of the cycle. The only staff concern is the degree of conservatism at the end of the cycle.

To resolve the staff's concern, TVA has agreed to reanalyze the transients at the end of cycle 2. The total cycle is estimated to result in 5415 MWD/t exposure to the fuel. As noted above, the only concern is with the later part of the cycle. The OLMCPRs proposed by TVA as a result of the REDY analysis are conservative for at least the initial 2000 MWD/t exposure during the fuel cycle. Therefore, the staff has proposed, and the licensee has accepted, that the proposed OLMCPRs of 1.24 for 8x8 fuel and 1.21 for 8x8R fuel will apply for the first 2000 MWD/t exposure in cycle 2; that is, from the beginning of the cycle (BOC) to BOC + 2000 MWD/t. During this period, TVA will submit a reanalysis and the staff will reevaluate the OLMPCRs for the balance of the cycle.

3.2.3 Thermal-Hydraulic Improvement Features

3.2.3.1 Prompt Recirculation Pump Trip

The prompt recirculation pump trip feature was described in Reference 9. The system uses line breakers between the motor-generator sets and the pump motors. This location provides the rapid reduction in pump speed necessary for the feature to be effective during the transient discussed in Section 2.2.2. The system is designed to be of quality consistent with the reactor protection system. The RPT design was reviewed and accepted for Cycle 2 of Browns Ferry Unit 2 (Reference 10). The design remains acceptable.

3.2.3.2 Simmer Margin

The licensee has proposed changes to the Technical Specifications which will increase the capacity (by installing larger valves) of the safety/relief valves from 78.7% to 84.2% of nuclear boiler rated (NBR) steam flow, and also increase the setpoints of the relief valves. (The safety valve capacity and setpoints were not changed.) The transient, overpressure, and LOCA analyses performed for the Cycle 2 analysis assumed this change.

The criterion for simmer margin is that only relief valves open during anticipated transients. Safety valves should not open under these conditions.

The analysis of the limiting pressure and power increase transient, which is the worst case for anticipated pressure events, predicted a pressure of 1203 psig at the safety valves, which is well below the 1250 psig safety valve setpoint. Moreover, peak pressures calculated with the REDY code have always been greater than those calculated using ODYN (Reference 8), and thus the concerns outlined in Section 3.2.2 do not apply here. Therefore, we find these changes to be acceptable.

3.3 Accident Analysis

3.3.1 ECCS Appendix K Analysis

Input data and results for the ECCS analysis have been given in Reference 1, 3, and 11. The information presented fulfills the requirements for such analyses outlined in Reference 6.

We have reviewed the analyses and information submitted for the reload and conclude that the BF-3 plant will be in conformance with all requirements of 10 CFR 50.46 and Appendix K to 10 CFR 50.46 when (1) it is operated within the "MAPLHGR VERSUS AVERAGE PLANAR EXPOSURE" values given in Tables 3.5.I-1, -2, and -3 of Reference 2, and (2) it is operated at a Minimum Critical Power Ratio (MCPR) greater than or equal to 1.20 (more restrictive MCPR limits are currently required for reasons not connected with the Loss of Coolant Accident, as described in Section 3.2.2).

3.3.2 Control Rod Drop Accident

For BF-3 Cycle 2, the generic scram reactivity curve (cold and hot) and the accident reactivity insertion curve (cold) do not satisfy the requirements for the bounding analyses described in Reference 5. Therefore, it was necessary for the licensee to perform plant and cycle specific analyses for the control rod drop accident for hot and cold startup conditions. The results of these analyses indicate that the peak fuel enthalpy for these events would be less than or equal to 280 calorics/gram, which is acceptable.

3.3.3 Fuel Loading Error

As discussed in Section 3.2.2, potential fuel loading errors involving misoriented bundles have been explicitly included in the calculation of the operating limit MCPR. Potential errors involving bundles loaded into incorrect positions have also been analyzed by a method which considers the initial MCPR of each bundle in the core, and the resultant MCPR was shown to be greater than 1.07. This GE method for analysis of misoriented and misloaded bundles has been reviewed and approved by the staff (Reference 12).

The analyses which have been performed for potential fuel loading errors for BF-3 Cycle 2 are acceptable for assuring that CPRs will not be below the safety limit MCPR of 1.07.

3.3.4 Overpressure Analysis

The overpressure analysis for the MSIV closure with high flux scram, which is the limiting overpressure event, has been performed in accordance with the requirements of Reference 6. As specified in Reference 8, the sensitivity of peak vessel pressure to failure of one safety valve has also been evaluated. We agree that there is sufficient margin between the peak calculated vessel pressure and the design limit pressure to allow for the failure of at least one valve. Therefore, the limiting overpressure event as analyzed by the licensee is considered acceptable.

2.3.5 ADS Out-of-Service Analysis

The automatic depressurization system (ADS) is provided to aid in vessel depressurization following a small break loss-of-coolant accident (LOCA). Thus, the ADS only affects the results of break analyses where depressurization through the break itself is relatively slow (small breaks), and operation of the ADS increases the depressurization rate, allowing low pressure systems (such as the core spray (CS) and the low pressure coolant injection (LPCI) systems) to reach higher flows sooner. This causes earlier reflood and lower calculated peak cladding temperature (PCT) results for the small break analyses. The more installed relief capacity (i.e., number of valves) in the ADS, the more pronounced is this effect.

Previous small-break analyses, in the small break size range where ADS has an appreciable effect (0 to approximately 0.5 ft^2), took credit for operation of five of the six ADS valves (Reference 13). Maximum PCT in that break size range was around 1530°F, far below the larger (and limiting) break sizes whose PCTs are around but still below 2200°F.

Continuous reactor operation with only four of the six ADS valves operable is acceptable if the small breaks' PCTs do not exceed 2200°F for any fuel operating at the MAPLHGR limit.

The application for change in the Technical Specifications (Reference 3) contained a generic estimate of a 200°F PCT increase for small breaks in the range affected by ADS capacity (0 to 0.5 ft^2). We have previously required substantiation of that estimate for Units 1 and 2 of Browns Ferry, which was provided in Reference 14 as discussed below. The results also apply to BF-3, as the three plants are similar except that BF-3 does not have the LPCI modification. The LPCI modification will have no effect on this analysis because loss of HPCI is the worst single failure.

(1) The estimate of 200°F PCT increase was provided for the Browns Ferry plants by a generic ADS out-of-service analysis, which included calculations for a 251-inch inside diameter pressure vessel (Reference 14). BF-3 is within this category.

(2) The generic estimate of 200°F PCT increase was confirmed for the ADS steam flow range appropriate for BF-3 (with four and five ADS valves operable) by the generic ADS out-of-service analysis, which included the BF-3 ADS' capacity range.

(3) The model used for the generic ADS out-of-service analysis did not contain the latest model changes described in Reference 15. However, those model changes have not caused significant changes in the PCT results for the small break analyses of a smaller sized BWR/4 and an identically sized BWR/3 (Reference 14), and similarly the changes would not significantly affect small break PCT results for BF-3.

For other reasons, the model changes (Reference 15) allowed operation at slightly higher MAPLHGR limits. At these higher powers, small break PCT results could be as much as 40°F higher. Therefore, PCT for the worst small break with four of the six ADS valves operable would be approximately 1460°F + 200°F + 40°F = 1700°F. This is considerably below 2200°F and is therefore acceptable.

We, therefore, conclude that the material presented and discussed above adequately supports the TVA request to operate continuously with four of the six ADS valves in service, and such operation is, therefore, acceptable.

3.3.6 Recirculation Pump Trip Failure

It is extremely unlikely that the RPT feature will fail. However, the consequences must be examined to see if they lie within the accident criteria.

The limiting pressure and power increase transient, with failure of the RPT feature, may result in fuel failure if all plant parameters are close to worst-case condition. Radioactive material could then be released through the feedwater pump turbines, steam jet air ejectors, and gland seals. (Most of this material would have to pass through the offgas system before release.) The specific activity within the steam would have to be below the value which would trigger MSIV closure on high steam line activity. An incident which caused isolation on high activity would be bounded by the analysis of the steam line break in the plant FSAR. Since the high steam line radiation setpoint is required by the Technical Specifications to be no more than three times normal background, transients coupled with RPT failures leading to coolant activities greater than three times the Technical Specification maximum would fall into this category.

During the course of the limiting pressure and power increase transient, the increasing water level reaches the high level set-point eight seconds into the transient, which trips the feedwater turbines. The water level then reaches a maximum and recedes. We estimate (by extrapolation of the data in Reference 1) that the level will drop to the low low setpoint after approximately 25 seconds. At this point, HPCI and RCIC initiate and the MSIV begin to close (Group I isolation). MSIV closure requires three to five additional seconds.

Failure of the RPT feature should not greatly affect the water level behavior except in the very early stages of the transient, when the void-sweeping effects are important. Once the MSIVs close, the radioactive releases will be bounded by the steam line break accident. Therefore, the important question is: how much steam flows through the feedwater turbines, steam jet air ejectors and gland seals in the 25 seconds before isolation?

At full power, the feedwater turbines on any LWR installation consume 2% or less of the main steam flow. The SJAEs and gland seals consume much less. Moreover, the feedwater turbines are tripped after eight seconds.

Clearly, assuming three times the maximum permissable coolant activity, 2% steam flow for eight seconds plus much less than 2% for 22 additional seconds will result in less release than 200% steam flow for five seconds at the maximum permissable coolant activity. The difference is greater than a factor of 5. Therefore, the 200%-five second assumptions of the steam line break analysis are bounding, and the consequences of RPT failure are acceptable.

Common mode failures must also be examined. The RPT feature operates off the same steam chest switches as the reactor scram on trip/fast closure. The licensee has referred (Reference 4) to probabilistic analyses submitted on other dockets. These analyses conclude that the probability of failure of the reactor scram is on the order of 10^{-6} per demand (Reference 16). The switches are only one contributor to this failure rate. Moreover, the RPT hardware is of similar quality to the reactor scram hardware (Reference 9). Therefore, it is concluded that the probability of simultaneous failure of the trip/fast closure scram and the RPT feature is much less than 10^{-6} per demand, and therefore need not be considered.

It is our judgement that all other simultaneous failures (e.g., caused by a seismic event) would necessitate failure of some equipment but not others in arrays which are of negligible probability.

3.4 Thermal Hydraulic Stability

The results of the thermal hydraulic stability analysis (Reference 1) show that the channel hydrodynamic and reactor core decay ratios at the natural circulation - 105% rod line interaction (which is the least stable physically attainable point of operation) are below the stability limit.

Because operation in the natural circulation mode at greater than 50% rated thermal power is prohibited by the Technical Specifications, there is added margin to the stability limit and this is acceptable.

3.5 Physics Startup Testing

The licensee will perform a series of physics startup tests and procedures to provide assurance that the conditions assumed for the transient and accident analysis calculations will be met during Cycle 2. The tests will check that the core is loaded as intended, that the incore monitoring system is functioning as expected, and that the process computer has been reprogrammed to properly reflect changes associated with the reload.

The licensee has stated (Reference 17) that the methods, criteria and reporting requirements for the test program will be, with two exceptions, identical to these accepted for Unit 2 (Reference 10).

The first exception involves the action to be taken in the event that the TIP asymmetry test indicates that the TIP instrumental uncertainty is in excess of that assumed in the development of the safety limit MCPR (Section 5 of Reference 5). Normally, an instrumental uncertainty higher than that assumed in the safety analyses would require additional safety margin, and thus some operating limit penalty.

TVA stated (Reference 17) that increased instrumental uncertainties will automatically penalize the operation of the plant in terms of MCPR, MAPLHGR, MLHGR and TPF by an amount greater than the penalty that would be calculated by a re-assessment of the safety limit assumptions. The reason this effect takes place is because (1) there are many locations in the core which run at powers very nearly equal to that of the peak power location, and (2) the operating limits are written in terms of maxims. Thus, even if the maximum location is read low due to instrumental uncertainty, there is a nearly unity probability that another location, almost as high in power, will be read high. Provided the peak location is accompanied by many other locations which are less in power by an amount which is much smaller than the instrumental uncertainty, the maximum value read by the incore instrumentation will automatically be conservative. Moreover, this automatic penalty rises in a nearly linear fashion as the instrumental uncertainty increases.

Since the instrumental uncertainty assumed in the safety analysis is combined statistically (i.e., RMS) with other allowances, the penalty calculated from the safety analysis rises less than linearly with increased instrumental uncertainty. Therefore, the automatic penalty discussed above is always greater than or equal to the appropriate safety penalty. Since the BF-3 Cycle 2 core meets all of the above criteria, we find this change to the startup test program to be acceptable.

The second exception involves the comparison of predicted vs. measured core power maps at high power, BOC conditions. The licensee has expressed difficulty in distinguishing power map discrepancies from instrumental noise and maintains that the balance of his testing program will detect any anomalies in the core (reference 17). Therefore, the licensee desires to eliminate this test.

After reviewing the licensee's core loading and past experience with power map uncertainties, we agree that this test is insufficiently sensitive to detect most postulated core anomalies. Moreover, examination of the presently available studies of the sensitivity of BWR core power maps to various perturbations indicates that there are not enough of these studies presently available to allow interpretation of core power maps discrepancies, even if such discrepancies could be unambiguously identified. Therefore, we find this second change to the startup program to be acceptable.

3.6 Rod Sequence Control System

Section 3.3.B.3.a of the present Technical Specifications for BR-3 contains a note which reads: "The Rod Sequence Control System (RSCS) has been evaluated only through the first refueling outage. A complete reevaluation is required prior to operations following the first refueling". As discussed in the introduction, BF-3 shutdown for the first refueling on September 8, 1978. BF-3 now has the Group Notch RSCS, as discussed in Reference 5 and accepted in Reference 6. Therefore, we find that the licensee's proposed deletion of the note in Section 3.3.B.3.a of the Technical Specifications is acceptable.

3.7 Primary Containment Isolation Valves

The surveillance requirements for testing primary containment integrity are specified in Section 4.7 of the Technical Specifications. Section 4.7.A.2.g states that local leak rate tests shall be performed on the primary containment testable penetrations and isolation valves at certain specific pressures and intervals. The testable penetrations and valves are listed in seven tables (3.7B thru 3.7H).

Table 3.7.D lists 105 primary containment isolation valves by number of the valve, the test medium to be used to test the specific valves (i.e., air or water) and the sections of lines to be tested for each valve (i.e., the test pressure will be applied, for example, between valves 74-48, 74-49 and 74-661). The inservice inspection and testing program for Browns Ferry has been under review by the staff and the licensee for the past two years (see TVA's submittals of May 25, 1977 and July 29, 1977, our letters of February 25, 1977 and August 8, 1978 and summary of meetings held August 15 and 16, 1978 between the staff and TVA on the ISI program). As a result of the continuing efforts to keep up with the Appendix J requirements, TVA has proposed to change the section of line to be tested for three of the 105 valves in Table 3.7.D (i.e., the hydrostatic test will be applied between different valves). The changes do not change the valves to be tested or the test medium to be used (water in all 3 cases). The changes are proposed to permit testing of more than one valve at a time.

Table 3.7.6 lists 15 check valves on drywell influent lines that are required to be tested. TVA proposes to delete the check valve that was listed for the control rod drive return line since it no longer exists in the plant; the CRD return line was rerouted and the penetration capped at the reactor vessel to reduce the potential for intergranular stress corrosion cracking. TVA also proposes to change the section of line to be tested for 6 of the check valves to eliminate testing each valve individually to reduce the initial test time. There are no proposed changes to the valves to be tested, other than for the CRD return line, and no change in the test medium.

The staff concludes that the proposed changes to the test procedures for the primary containment isolation and check valves are in accordance with 10 CFR 50, Appendix J, they do not in any way change the valves to be tested and that the proposed changes are acceptable.

4.0 Environmental Considerations

We have determined that these amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that these amendments involve an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 51.5(d)(4) that an environmental impact statement, or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

5.0 Conclusion

We have concluded: (1) because the amendments do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease in a safety margin, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of these amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: NOVEMBER 1 8 1978

6.0 References

1. Supplemental Reload licensing submittal for Browns Ferry Nuclear Plant Unit 3 Reload 1, NEDO-24128, June, 1978, submitted as Enclosure 2 of letter, O. E. Gray III (TVA) to H. R. Denton (NRC), dated August 3, 1978.

2. Enclosure 1 of leter, O. E. Gray III (TVA) to H. R. Denton (NRC), dated August 3, 1978.

3. Loss-of-Coolant Accident Analysis for Browns Ferry Nuclear Plant Unit 3, NEDO-24127, June, 1978, submitted as Enclosure 2 of letter, O. E. Gray III (TVA) to H. R. Denton (NRC), dated August 3, 1978.

4. Letter, J. E. Gilleland (TVA) to Director of Nuclear Reactor Regulation (NRC), dated October 20, 1978.

5. General Electric Boiling Water Reactor Generic Reload Application, NEDE-24011-P, May, 1977.

6. Safety Evaluation for the General Electric Topical Report "Generic Reload Fuel Application," NEDE-24011-P, April, 1978, transmitted as enclosure of letter, D. G. Eisenhut (NRC) to R. Gridley (GE), dated May 12, 1978.

7. Analytical Methods of Plant Transient Evaluations for the General Electric Boiling Water Reactor, NEDO-10802, February, 1973.

8. Impact of One-Dimensional Transient Model on Plant Operations Limits, enclosure of letter, E. D. Fuller (GE) to U. S. Nuclear Regulatory Commission, dated June 26, 1978.

9. Basis for Installation of Recirculation Pump Trip System, NEDO-24119, April, 1978.

10. Safety Evaluation by the Office of Nuclear Reactor Regulation Supporting Amendment No. 35 to Facility Operating License No. DPR-52, enclosed in letter, T. A. Ippolito (NRC) to TVA (Attn. N. B. Hughes), dated June 21, 1978.

11. Letter, Lee Liu (Iowa Electric Light & Power Co.) to Edson G. Case (NRC), Letter No. IE-77-1453, dated July 29, 1977.

12. Letter, D. G. Eisenhut (NRC) to R. Engel (GE), dated May 8, 1978.

13. Letter, J. Gilleland (TVA) to E. Case (NRC), dated October 28, 1977.

14. Letter, J. E. Gilleland (TVA) to Director of Nuclear Reactor Regulation (NRC), dated April 20, 1978.

15. Letter, K. Goller (NRC) to G. Sherwood (GE), SER for GE ECCS Evaluation Model, dated April 12, 1977.

16. 251 NSSS GESSAR, Attachment A to Response to Staff Question 222.22.

17. Minutes of meeting between NRC staff and TVA, held in Bethesda on September 26, 1978.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-259, 50-260, AND 50-296

TENNESSEE VALLEY AUTHORITY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 45 to Facility Operating License No. DPR-33, Amendment No. 41 to Facility Operating License No. DPR-52 and Amendment No. 18 to Facility Operating License No. DPR-68 issued to Tennessee Valley Authority (the licensee), which revised Technical Specifications for operation of the Browns Ferry Nuclear Plant, Units Nos. 1, 2 and 3, located in Limestone County, Alabama. The amendments are effective as of the date of issuance.

Amendment No. 18 changes the Technical Specifications to incorporate the limiting conditions for operation associated with the initial 2000 megawatt davs per tonne (MWD/t) fuel exposure during the second fuel cycle for Unit No. 3. The amendments also incorporate minor changes in the test setups to be used to test certain primary containment isolation and check valves.

The application for the amendments complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendments. Prior public notice of these amendments was not required since the amendments do not involve a significant hazards consideration.

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The Commission has determined that the issuance of these amendments will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of these amendments.

For further details with respect to this action, see (1) the application for amendments dated August 3, 1978, as supplemented by letter dated October 20, 1978, (2) Amendment No. 45 to License No. DPR-33, Amendment No. 41 to License No. DPR-52, and Amendment No. 18 to License No. DPR-68, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Athens Public Library, South and Forrest, Athens, Alabama 35611. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 18th day of November 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

T. A. Ippolito

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors